WITHHOLD ENCLOSURE 3 FROM PUBLIC DISCLOSURE
UNDER 10 CFR 2.390 and 9.17

August 21, 2009

L-MT-09-044
10 CFR 50.90

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

Monticello Nuclear Generating Plant
Docket 50-263
Renewed Facility Operating License
License No. DPR-22

Monticello Extended Power Uprate: Response to NRC Mechanical and Civil Engineering Review Branch (EMCB) Requests for Additional Information (RAIs) dated March 28, 2009 (TAC MD9990)

References:
1. NSPM letter to NRC, License Amendment Request: Extended Power Uprate (L-MT-08-052) dated November 5, 2008, (TAC MD9990) Accession No. ML083230111
2. Email P. Tam (NRC) to G. Salamon, K. Pointer (NSPM) dated March 28, 2009, "Monticello - Draft RAIs from Mechanical & Civil engineering Branch re: proposed EPU amendment (TAC MD9990)" Accession No. ML090880002

Pursuant to 10 CFR 50.90, the Northern States Power Company, a Minnesota corporation (NSPM), requested in Reference 1 an amendment to the Monticello Nuclear Generating Plant (MNGP) Renewed Operating License (OL) and Technical Specifications to increase the maximum authorized power level from 1775 megawatts thermal (MWt) to 2004 MWt.

On March 28, 2009, the U.S. Nuclear Regulatory Commission (NRC) Mechanical and Civil Engineering Review Branch (EMCB) provided the requests for additional information (RAIs) contained in Reference 2. Enclosure 1 provides the proprietary response to EMCB RAIs in References 2. A non proprietary version of Enclosure 1 is contained in Enclosure 3. GEH requests this proprietary information to be withheld from public disclosure in accordance with 10 CFR 2.390(a)4 and 9.17(A)4. An affidavit supporting this request is provided in Enclosure 2. Enclosure 4 is provided for information.

In accordance with 10 CFR 50.91, a copy of this letter is being provided to the designated Minnesota Official without the proprietary version.

Summary of Commitments

1. Confirmation that Feedwater and Condensate pump and heater replacement modifications are complete and meet the code allowables will be provided to the NRC prior to implementation of the EPU license amendment request.

2. Confirmation that modification of support TWH-143 is complete will be provided to the NRC prior to implementation of the EPU license amendment request.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on August 21, 2009.

for TJ O'Connor

Timothy J. O'Connor
Site Vice President, Monticello Nuclear Generating Plant
Northern States Power Company - Minnesota

Enclosure

cc: Administrator, Region III, USNRC
Project Manager, Monticello, USNRC
Resident Inspector, Monticello, USNRC
Minnesota Department of Commerce

ENCLOSURE 3

NSPM RESPONSE TO EMCB RAIs DATED MARCH 28, 2009

Non Proprietary

EMCB RAI No. 1

Provide a table which contains information on plant operating parameters similar to Table 1-2 and include a column for OLTP. Include design and maximum temperatures for reactor recirculation system (RRS) vessel outlet and inlet nozzles and feedwater (FW) nozzles.

NSPM RESPONSE

Plant Operating Conditions	OLTP	CLTP[1]	EPU
Thermal Power (MWt)	1670	1775	2004
Vessel Steam Flow (Mlb/hr)	6.78	7.26	8.34
Full Power Core Flow Range			
Mlb/hr	43.2 to 60.5	47.5 to 60.5	57.0 to 60.5
% Rated	75 to 105	82.4 to 105	99.0 to 105
Maximum Normal Dome Pressure (psia)	1025	No Change	No Change
Maximum Normal Dome Temperature (ºF)	548	No Change	No Change
Pressure Upstream of TSV (psia)	965	970	952
Full Power Feedwater			
Flow (Mlb/hr)	6.75	7.24	8.31
Temperature (ºF)	377	383.0	395.8
Core Inlet Enthalpy (Btu/lb)[2]	524.6	523.7	523.0

1. Based on current reactor heat balance; 2. At 100% core flow condition

Reactor Nozzle	OLTP	CLTP	EPU Value
RRS Outlet Design Temperature	575°F	No Change	No Change
RRS Outlet Maximum Temperature[1]	546°F	549°F	548°F
RRS Inlet Design Temperature	575°F	No Change	No Change
RRS Inlet Maximum Temperature[1]	546°F	549°F	548°F
FW Nozzle Design Temperature	575°F	No Change	No Change
FW Nozzle Maximum Temperature	376°F	385°F	398°F

1. Maximum temperature is saturation temperature for reactor with no feedwater flow assumed. OLTP value is based on normal reactor pressure of 1000 psig, CLTP value is based on normal reactor pressure of 1025 psig and EPU value is based on normal reactor pressure of 1010 psig.

EMCB RAI No. 2

Confirm whether the current licensing basis criteria for high energy line break (HELB) are the criteria contained in the Giambusso/Schwencer letters (1972-73).

NSPM RESPONSE

These criteria were not changed for EPU. USAR Appendix I, Section I.1, defines the evaluation criteria for HELBs. The USAR states:

"The criteria used for the determination of the high energy lines and the effects of the postulated breaks on these lines on safe shutdown equipment are the December 18, 1972 Giambusso letter (Reference 2) as clarified by Standard Review Plan (SRP) 3.6.1 (Reference 3), SRP 3.6.2 (Reference 4), and Generic Letter 87-11 (Reference 21). These criteria are utilized as the basis for the determination of the high energy lines, break locations, and the evaluation of effects on Safe Shutdown (SSD) equipment."

The associated USAR references are:

2. Letter from A. Giambusso, Deputy Director for Reactor Projects, to Northern States Power Company, Subject: High Energy Breaks Outside of the Containment, December 18, 1972.
3. Standard Review Plan 3.6.1, Plant Design for Protection Against Postulated Piping Failures in Fluid Systems Outside Containment, Rev. 1, July 1981.
4. Standard Review Plan 3.6.2, Determination of Rupture Locations and Dynamic Effects Associated with the Postulated Rupture of Piping, Rev. 1, July 1981.
21. NRC (F J Miraglia) Generic Letter 87-11, "Relaxation in Arbitrary Intermediate Pipe Rupture Requirements", June 19, 1987.

Staff review and acceptance of the analyses performed and the measures taken in response to the December, 1972 letter from A Giambusso is documented in the July 29, 1974 letter from Karl R Goller to Northern States Power Co. Letter, AEC to NSP, United States Atomic Energy Commission - Safety Evaluation by the Directorate of Licensing, Docket No. 50-263, Monticello Nuclear Generating Plant - "Analysis of the Consequences of High Energy Piping Failures Outside Containment"

Documentation of further Staff review is provided in letter, NRC to NSP, Monticello - High Energy Line Break Analysis (TAC No. 61788), June 13, 1990.

EMCB RAI No. 3(a)

PUSAR Section 2.2.1 states that corrective actions are underway to perform HELB analysis upgrades at Monticello due to changes in pipe break methodology.

Explain why corrective actions are in place to upgrade the Monticello pipe break methodology.

NSPM RESPONSE

PUSAR Section 2.2.1 states:

Technical Evaluation

No changes to the implementation of the existing criteria for defining pipe break and crack locations and configurations are being made for EPU . . .

Changes in Methods of Analysis

The results provided for HELB events affected by EPU, specifically, the liquid line breaks in the Feedwater, Condensate, and RWCU systems show much larger changes than would be expected due to the small changes in pump discharge pressures and small enthalpy changes as a result of EPU. The results are driven by conservative changes in analysis methods resulting from corrective actions underway to perform HELB analysis upgrades at Monticello."

The criteria used to determine high energy lines has not changed with EPU, see RAI 2 above. The changes NSPM referred to are covered in corrective action program action request AR01131913, HELB Program Documentation Deficiencies, which documents a summary of issues being addressed. The most significant changes are related to the assumptions used in determining mass and energy releases from postulated breaks and upgrade of the computer code from GOTHIC version 4.0. The EPU liquid break calculation inputs have been upgraded to consider:

1. Double-ended break flow to include flow from both ends of postulated breaks
2. System depletion to include mass and energy that exists in system piping and pressure vessels
3. A conservative change in assumption for isolation valve stroke time from ASME Section XI Limiting Stroke time to the value listed as the maximum valve operating time in the USAR. If break detection logic exists, valve stroke is initiated when the logic detects the break.
4. A conservative change for flow reduction assumptions with valve closure. CLTP analysis assumed flow was reduced proportional to isolation valve percent closed position. The EPU analysis assumed 100% break flow until isolation valve was fully closed.

5. The liquid mass from fire protection sprinkler systems postulated to actuate from HELB events was included
6. Upgrade computer code from GOTHIC version 4.0 to GOTHIC version 7.1 or later versions

   The assumption changes noted above are based on recommendations from site self assessments. These changes will bring the HELB program into closer alignment with industry standards and correct identified deficiencies. The failure to consider fire protection sprinkler system actuation for appropriate HELBs resulted in the issuance of LER 2008-001, Non-Conservative HELB Analysis discovered during EPU, and is documented under AR1125675.

   Re-analysis of all HELB breaks and an evaluation of affected EQ components have been completed; formal updating of EQ program documents are the only actions remaining. These actions are being performed coincident with EQ program updates required by EPU.

EMCB RAI No. 3(b)

Verify whether the Monticello pipe break methodology upgrade is based on SRP Section 3.6.2, MEB 3-1 criteria. If not, provide supporting justification.

NSPM RESPONSE:

As noted above in the response to Part a) of this question, there is no change to the pipe break methodology at Monticello. The changes involve a re-analysis of breaks using more conservative assumptions of mass and energy release.

EMCB RAI No. 4

ELTR 1 and ELTR 2 both recommend that HELB evaluations for High Pressure Core Spray (HPCS) and Building Heating Line be performed on plant-specific power uprate submittals. Please indicate where in the proposed LAR submittal these evaluations have been performed or provide the HELB plant-specific evaluations for these systems at EPU conditions.

NSPM RESPONSE

Monticello does not have a High Pressure Core Spray system, see USAR Section 6.2.

The criteria for HELB consideration at Monticello are for piping systems that are >275 psig and >200°F, see USAR Appendix I.2. This is based on United States Atomic Energy Commission - Safety Evaluation by the Directorate of Licensing, Docket No. 50-263, Monticello Nuclear Generating Plant - "Analysis of the Consequences of High Energy Piping Failures Outside Containment", July 29, 1974 (Enclosure 4). Building heating lines at Monticello do not meet criteria for consideration under the HELB program and therefore were not evaluated.

EMCB RAI No. 5

Page 2-23 states that:

> "During the 6.3 percent rerate in 1996, only one new case was reanalyzed at CLTP for the RWCU system - a break in the system suction piping at the outboard isolation valve. For this reason a detailed comparison of CLTP and EPU results for HELBs in the RWCU system is not possible."

The statement that, "For this reason a detailed comparison of CLTP and EPU results for HELBs in the RWCU system is not possible" is not clear. Please provide clarification.

NSPM RESPONSE

The CLTP analysis of RWCU HELBs evaluated the terminal end break and crack case at the inlet to the RWCU heat exchanger. The evaluation used the mass and energy release rates for a break just outboard of the outboard isolation valve. These were considered the bounding cases and other cases were not run. For EPU, eight HELB locations, covering all possible breaks and cracks, were evaluated.

The response to RAI 3.a above explains changes in assumptions used in evaluation of the EPU HELB cases. As noted on PUSAR page 2-21:

> Because of these changes in methodology, a comparison of the results between EPU and CLTP conditions shows a significantly larger change than would normally be expected based on the small changes in process fluid temperatures and enthalpy resulting from EPU based on previous industry experience.
>
> Monticello has chosen not to perform a full re-analysis of these specific liquid line HELBs at CLTP conditions because it was determined that our effort should be focused on completing the corrective actions using bounding conditions. Thus, a detailed breakdown of the magnitude of the change is caused by EPU versus the change resulting from the changes in methods and correction of errors is not provided.

A comparison of the results between EPU and CLTP conditions was not done since it would have required the creation of an additional 12 calculations to define CLTP conditions with the new assumptions included. This significant effort was not warranted as the bounding analysis completed for EPU have addressed the desired CLTP analysis improvements. Results of a comparison between the single CLTP RWCU HELB case and the similar EPU HELB case is discussed in RAI 6 below. Re-analysis of all HELB breaks and an evaluation of affected EQ components have been completed; formal updating of EQ program documents are the only actions remaining. These actions are being performed coincident with EQ program updates required by EPU.

EMCB RAI No. 6(a)

The same paragraph on page 3-23, as above, in reference to the reactor water cleanup (RWCU), continues as follows:

> "For the break location that was analyzed during Rerate, new mass and energy release calculations considered additional blowdown sources that had not been considered in the previous 1996 analysis. This resulted in an increase in integrated mass release of about 90% and an increase in integrated energy release of 63 percent."

Confirm that the 90% and 63% increases are referring to the proposed EPU.

NSPM RESPONSE:

The 90% and 63% increases are not referring to the proposed EPU. It is referring to the change in assumptions as noted in response to RAI 3 above rather than system operating condition changes resulting from EPU.

If the CLTP HELB cases were run using similar assumptions, the changes in mass and energy releases would be minor as a result of EPU.

As noted on PUSAR page 2-21:

> A review of the results from several recent EPU submittals concluded that, in most cases, environmental conditions are bounded by previous analyses, confirming that EPU produces relatively minor effects.

EMCB RAI No. 6(b)

Please explain how the effects of the increased mass and energy release have been evaluated, include evaluations of pipe whip restraints and jet targets.

NSPM RESPONSE

Changes in mass and energy were evaluated for impacts on HELBs using the GOTHIC code. This allowed a determination of time histories for all plant areas to evaluate effects on temperature, pressure and flooding. Differential pressures between plant areas verified acceptable margins for structures such as block walls. The effects of changes to temperature, pressure and flooding have been evaluated for impact on the environmental qualification (EQ) of equipment. Upgrades to EQ files to document this evaluation are in progress.

RWCU pipe whip, jet impingement and safe shut down analyses following postulated pipe breaks or cracks are provided in USAR Appendix I. The RWCU high energy lines are located in the RWCU compartment, steam chase; MG set room, and the North West side of elevations 962' and 935' of the reactor building. There are no postulated breaks in the MG set room and the reactor building elevations 962' and 935' based on seismic analysis. There are no pipe whip targets for the RWCU piping in the steam chase.

The safe shutdown evaluation for the RWCU compartment in Appendix I does not rely on pipe whip restraints or jet impingement shields to protect any equipment or structures. The effects of pipe whip and jet impingement in this area do not result in the loss of components required to mitigate the break and shut down the reactor. Therefore there is no impact on RWCU pipe whip and jet impingement due to EPU.

EMCB RAI No. 7

Page 2-37 states that: "The combination of stresses was evaluated to meet the requirements of the pipe break criteria. Based on these criteria, no new postulated pipe break locations were identified." For systems affected by the EPU, specifically steam (all EPU affected steam lines) and FW lines (including condensate), provide a pipe break analysis summary table (that includes the main steam increased turbine stop valve (TSV) closure transient loads in the analysis) which compares values at EPU and CLTP conditions and shows code equation stresses and CUFs compared to break limit for stresses and CUFs. Include pipe break locations and types selected for CLTP and EPU. Include lines inside and outside containment.

NSPM RESPONSE

Systems that have piping meeting the MNGP design basis criteria for classification as "High Energy" include Main Steam, Condensate, Feedwater, Residual Heat Removal (RHR), Core Spray (CS), High Pressure Coolant Injection (HPCI), Reactor Core Isolation Cooling (RCIC), Reactor Water Cleanup (RWCU), Off Gas, Control Rod Drive (CRD), Zinc Oxide Injection (GEZIP), and Standby Liquid Control (SLC). The parameters used for stress analysis in the high energy portions of these systems are unchanged due to EPU except in the Main Steam, Condensate, Feedwater, and GEZIP systems.

The Main Steam system analysis results including TSV closure loads are provided in the table below. The stress result for the Main Steam location with the maximum HELB break postulation equation result is also included in the table. The stress at that location does not meet (is less than) the current design basis criteria to require a postulated break. Hence, there is no Main Steam break outside containment postulated based on stress criteria. Other postulated break locations are based on configuration (e.g., terminal ends) which is not changed by EPU. Note that in the current design basis, specific HELB locations are not postulated inside containment. The current design basis does not include fatigue analysis of the Main Steam piping. Due to the revised analysis of the turbine stop valve closure loads, comparison to pre-EPU values is not meaningful.

The Main Steam evaluation results shown below are performed for the EPU pressure, temperature and flow parameters, including the TSV closure loads.

Main Steam Outside Containment - Maximum EPU Results (Highest Interaction Ratio):

Load Combination	Service Level	Node	Stress psi	Allowable psi	Ratio S/Allow
P+DW	B	X7A	6877	15000	0.46
TH Range	B	TURB	19441	22500	0.86
P+DW+TSV	B	268	12236	18000	0.68
DW+TSV+SRV+SSE	D	268	13795	26325	0.52
HELB DW+TH+OBE	B	TURB	27559	30000	0.92

The maximum Feedwater system operating temperature is 397.7^{o}F at EPU conditions for the Feedwater piping from the outboard containment isolation valve to the containment and inside containment. This value is bounded by the original analysis temperature of 400^{o}F. The design pressure for this portion of the Feedwater system is unchanged by EPU. Therefore this piping is unaffected by EPU relative to HELB postulation.

The feedwater piping and condensate piping from the condensate pump suction to the containment isolation valves will be re-analyzed during the Feedwater and Condensate pump and heater replacement modification process. High Energy Line Breaks and pipe whip restraints in the high energy portion of this piping will be evaluated at that time. GEZIP connections to the portion of the Feedwater system will be analyzed as part of the modification process. Details of the modifications to this piping are not yet finalized. The design will maintain stresses in the condensate and FW piping within code allowable limits of ANSI-B31.1-1977, including Winter 1978 Addenda and the requirements of USAR Chapter 12 including USAR Appendix I. Confirmation that the modifications are complete and meet the code allowables will be provided to the NRC in a separate letter. The FW and condensate system modifications are scheduled for completion during RFO25 in 2011.

EMCB RAI No. 8

Enclosure 5, PUSAR Section 2.2.1.2, Liquid Line Breaks, on page 2-23 states that:

"The mass and energy releases for HELBs in the RWCU, FW, Condensate, CRD, Standby Liquid Control, and Zinc Injection (GEZIP) systems and instrument and sample lines may be affected by EPU and were re-evaluated at EPU conditions. [[ ]] evaluations of liquid line breaks have been performed at EPU conditions."
Provide similar summaries as in RAI 7 for the RWCU line breaks at EPU conditions.

NSPM RESPONSE

From a HELB postulation viewpoint, there is no change in RWCU piping analysis temperature or design pressure due to EPU. Consequently, the pipe break postulation stress evaluations for RWCU are not changed at EPU conditions. Changes in mass and energy release are primarily due to the change in assumptions identified in response to RAI 3a above.

EMCB RAI No. 9

Indicate whether the FW lines have been structurally analyzed for any flow instabilities and loads due to water hammer or other flow transients and whether reanalysis has considered the EPU higher flows for these transients in evaluating pipe stresses, pipe breaks and pipe supports.

NSPM RESPONSE

The current analysis of the FW lines contains no structural analysis for any flow instabilities or loads due to water hammer or other flow transients. Such analysis was not performed at EPU conditions.

EMCB RAI No. 10

Are there any new liquid or steam line pipe break locations that need to be postulated due to EPU conditions?

NSPM Response

There are no new liquid or steam line pipe break locations that need to be postulated due to the change in process conditions at EPU.

Systems that are reconfigured by plant modifications (e.g., condensate and feedwater piping as identified in response to RAI 7) are evaluated during the modification process for HELB break postulation.

EMCB RAI No. 11(a)

For main steam (MS) and FW piping, state the design basis (DB) code for Class I and Class II piping and pipe supports.

EMCB RAI No. 11(b)

Verify that all structural evaluations of SSCs, required for EPU, were performed in accordance with the DB codes of record for piping and pipe supports. If a different code than the DB code of record was used, provide a justification.

NSPM Response

a. The MS piping system and associated branch piping (inside containment) were evaluated for compliance with the ASME Section III, Division I, 1977 Edition with Addenda up to and including Winter 1978 Piping Code stress criteria, including the effects of EPU on piping stress, piping supports including the associated building structure, piping interfaces with the RPV nozzles, containment penetrations, flanges, and valves. The requirements of ANSI B31.1-1977 through the W1978 addenda are used for FW piping and supports.

b. All structural evaluations of SSCs, required for EPU, were performed in accordance with the DB codes of record for piping and pipe supports as indicated on page 2-36 of the PUSAR.

EMCB RAI No. 12

Page 2-36 of the PUSAR states that, "The effects of the EPU conditions have been evaluated for the following piping [BOP] systems:" A list of piping systems follows this statement. On page 2-37 of the PUSAR, it is stated that, "These piping systems have been evaluated using the process defined in Appendix K of ELTR1 and found to meet the appropriate code criteria for the EPU conditions," when in fact evaluations of many of these systems, including RHR and MS, has not been completed, as shown by the submitted EPU LAR, see PUSAR Table 2.2-2d. In addition, Enclosure 8, Table 8-2 states that EPU planned modifications include, "Revise documentation to incorporate revised pressure and temperature ratings for specific piping systems affected by EPU. Modify supports as required by the analyses."

EMCB RAI No. 12(a)

The above PUSAR statements are not consistent. Please clarify the apparent inconsistency.

EMCB RAI No. 12(b)

The proposed EPU LAR indicates that some EPU evaluations have not been completed for the staff to review. The acceptability of the proposed EPU LAR will be determined based upon the results of the LAR evaluation reviews that are performed by the staff in accordance with the policies and procedures set forth in LIC-101, "License Amendment Review Procedures." Please provide a schedule of completion of these analyses and submittal of your evaluation results which shows that piping and pipe supports meet code allowable. Also, submit a schedule of completion for EPU required piping and pipe support modifications.

NSPM Response

Response to Part a

The referenced statement on page 2-37 indicating that pipe systems meet code requirements is intended to apply to piping stresses. Later on the same page, under the heading of "Pipe Supports," the structures listed on Table 2.2-2d are discussed. Based on the ELTR1 Appendix K methodology, the components listed were found to exceed code limits. Further, more detailed analysis may resolve some of these issues; others may require modification. This is consistent with the referenced statement from EPU LAR Enclosure 8 which indicates "supports" being modified as required by analysis.

Response to Part b

All piping and support evaluations required in ELTR1 have been completed using the methodology of Appendix K or by a more detailed analytical method. Completion of piping support detailed analysis and/or modifications for items listed in Table 2.2-2d was scheduled for the 2009 outage RF024. The current status of work shown on PUSAR Table 2.2-2d is provided below:

Table 2.2-2d Piping Components Requiring Further Reconciliation

Item	System	Current Status
1	Main Steam (Outside Containment)	Refined analysis is complete, all piping components and supports meet code allowables.
2	Feedwater and Condensate (from condensate pump to the feedwater MO valves downstream of the HP Heaters), due to pending pump changes	Replacement of feedwater heaters, condensate and feedwater pumps will result in nozzle changes that will impact piping layout and analysis. Final design of these components is still in progress and is scheduled for completion in the 2011 refueling outage. NSPM will complete piping analysis and modifications as noted in response to RAI 7.
3	Torus Attached Piping	Refined analysis is complete, all modifications are complete with exception of one support, TWH-143, which will be completed on-line prior to implementation. Confirmation that modification of support TWH-143 is complete will be provided to the NRC prior to implementation of the EPU license amendment request.
4	RHR (BOP Condensate Service Water Lines)	Refined analysis is complete, all piping components and supports meet code allowables.
5	Cross Around Piping	Replacement of CARVs and CARV discharge piping during the RFO impacted this analysis. Prompt evaluations of field changes from this work are complete and all piping and supports meet code allowables.

Item	System	Current Status
6	CARV Discharge Piping	Replacement of CARVs and CARV discharge piping during the RFO impacted this analysis. Prompt evaluations of field changes from this work are complete, all piping and supports meet code allowables.

EMCB RAI No. 13

a) Provide a list of systems (inside and outside containment) for which temperature, pressure, flow and mechanical loads have been increased due to EPU. Please include OLTP and EPU values.

b) Provide a brief summary that shows the EPU maximum code equation stresses compared to CLTP for the affected systems. For MS, FW and condensate see RAI 17, below. Include fatigue evaluation CUFs, where applicable. It is noted, that although the tables in Section 2.2 of the PUSAR include, for some BOP systems, the percentage increases for pipe stresses and pipe support loads, varying from 9 to 72 percent increases, due to temperature or pressure increases, these percentages are not indications that piping and pipe supports meet code equation allowable values, without providing maximum resulted values compared to code allowable.

NSPM Response

The system temperature, pressure, and flow changes due to EPU that are not bounded by the parameters used in the existing stress analyses are shown in Table 1, below.

The maximum code equation stresses for Main Steam at EPU conditions are summarized in Table 2, below. The maximum code equation stresses for BOP systems are summarized in Table 3, below. The current design basis does not include fatigue analysis of the Main Steam piping.

Table 1
MNGP EPU Piping Analysis Input Parameter Changes

Item	Parameter	OLPT	CLTP Value	EPU Value
Inside Containment				
1	**Main Steam**			
	Flow (Lbm/hr)	6.78E+6	7.262E+6	8.524E+06
2	**Feedwater,** from outboard containment isolation valves (FW-91-1 and FW-91-2) to RPV			
	Flow (Lbm/hr)	6.83E+6	7.313E+06	8.575E+06
3	**Core Spray** (CS)			
	Temperature (˚F)	180	196.7	212
Outside Containment				
1	**Main Steam**, upstream of TSV			
	Flow (Lbm/hr)	6.78E+6	7.262E+06	8.524E+06
2	**Feedwater,** From MO-1614/1615 to FW-91-1/FW-91-2			
	Flow (Lbm/hr)	6.75E+6	7.235E+6	8.497E+06
	From pumps to MO-1614 and MO-1615			
	Temperature (˚F)	400	400	Note 1
	Pressure (psig)	1550	1550	Note 1
	Flow (Lbm/hr)	6.75E+6	7.235E+6	8.497E+06
3	**Condensate**, from Condensate pump suction to Feedwater pump			
	Temperature (˚F)	302	310	Note 1
	Pressure (psig)	434	434	Note 1
	Flow (Lbm/hr)	6.75E+6	7.235E+6	8.497E+06
4	**Torus Attached Piping** (CS, HPCI, RCIC, Note 2)			
	Temperature (˚F)	180	196.7	212

Table 1
MNGP EPU Piping Analysis Input Parameter Changes

Item	Parameter	OLPT	CLTP Value	EPU Value
5	**Emergency Service Water**, ECCS Pump Room Ventilation Units (V-AC-4/5) Outlet Lines			
	Temperature (˚F)	120	120	122
6	**Extraction Steam**			
	Operating Temperature (˚F)			
	To Heater E-11	177	183	186
	To Heater E-12	236	246	253
	To Heater E-13	313	315	323
	To Heater E-14	344	348	358
	To Heater E-15	386	396	407
	Design pressure (psig)			
	To Heater E-11	-8	-7	-6
	To Heater E-12	8	13	16.8
	To Heater E-13	66	68	79
	To Heater E-14	111	117	136.4
	To Heater E-15	197	220	254
	Flow (Mlbm/hr)			
	To Heater E-11	0.404	0.592	0.700
	To Heater E-12	0.371	0.423	0.490
	To Heater E-13	0.443	0.444	0.525
	To Heater E-14	0.806	0.893	1.164
	To Heater E-15	0.388	0.443	0.548
7	**Heater Drains**			
	Operating Temperature (˚F)			
	From Heater E-11	173	180	183
	From Heater E-12	236	243	250
	From Heater E-13	241	248	254
	From Heater E-14	315	318	327
	From Heater E-15	343	349	359

Table 1
MNGP EPU Piping Analysis Input Parameter Changes

Item	Parameter	OLPT	CLTP Value	EPU Value
	Design pressure (psig)			
	From Heater E-11	-8	-7	-6.6
	From Heater E-12	7	12	15
	From Heater E-13	54	64	74
	From Heater E-14	96	110	128
	From Heater E-15	184	215	238
	Flow (Mlbm/hr)			
	From Heater E-11	2.52	2.80	3.43
	From Heater E-12	2.04	2.20	2.73
	From Heater E-13	1.67	1.78	2.24
	From Heater E-14	1.22	1.34	1.71
	From Heater E-15	0.39	0.44	0.55
8	**Service Water**			
	Inlet Temperature (˚F)	85	90	90
9	**Cross Around**			
	Temperature (˚F)	387	393	407
	Pressure (psig)	197	214	254
	Flow (Mlbm/hr)	6.33	6.75	7.91
10	**Cross Around Relief Valve Inlet**			
	Temperature (˚F)	381	389	403
	Pressure (psig)	182	204	242
	Flow	5.66	6.05	7.03
11	**Moisture Separator Drain**			
	Temperature (˚F)	383	392	403
	Pressure (psig)	202	204	242
	Flow (Mlbm/hr)	0.6728	0.7011	0.877

Note:
1. Due to the planned extensive piping modification to the Condensate and Feedwater systems, this piping is analyzed for EPU condition changes as part of the modification process (Reference response to RAI 7).
2 Torus attached RHR piping is currently analyzed at a temperature higher than the peak torus temperature and is therefore not changed by EPU.

Table 2
MNGP EPU Main Steam Piping and Support Results Summary

The Main Steam evaluation results shown below are performed for the EPU pressure, temperature and flow parameters, including the TSV closure loads.

Main Steam Inside Containment - Maximum EPU Results (Highest Interaction Ratio)

Maximum Pipe Stresses

Load Combination	Service Level	Node	Stress psi	Allowable psi	Ratio S/Allow
P+DW	B	161	7709	15000	0.51
TH Range	B	203	22940	22998	1.00
P+DW+OBE	B	U08	17823	18000	0.99
DW+TSV+SRV+SSE	D	U08	31261	36000	0.87

Note: 1. High Energy Line Breaks locations are not postulated inside containment.
2. Due to the revised analysis of the turbine stop valve closure loads, comparison to pre-EPU values is not meaningful.

Maximum SRV Flange Loads
Inlet Flange

Load Condition	Service Level	Node	Moment ft-lb	Allowable ft-lb	Ratio M/Allow
DW + TH	B	U07	14558	34083	0.427
DW + TH + Level B Dynamic	B	U07	39362	68167	0.577
DW + TH + Level D Dynamic	D	U07	65909	99750	0.661

Outlet Flange

Load Condition	Service Level	Node	Moment ft-lb	Allowable ft-lb	Ratio M/Allow
DW + TH	B	U08	13663	31000	0.441
DW + TH + Level B Dynamic	B	U08	34907	62083	0.562
DW + TH + Level D Dynamic	D	U08	57547	91250	0.631

Table 2
MNGP EPU Main Steam Piping and Support Results Summary

Main Steam Inside Containment - Maximum EPU Results (Highest Interaction Ratio)

Maximum RPV Nozzle Loads
RPV Nozzle N-3D

Loads	Service Level	Node	Fx lb	Fy lb	Fz lb	Mx ft-lb	My ft-lb	Mz ft-lb
Maximum Loads	B	101	6667	18555	4979	67422	18193	98764
Allowables	B	101	19392	51712	19392	258562	32320	258562
Maximum/Allowable	B	101	0.344	0.359	0.257	0.261	0.563	0.382

Maximum Flue Head Anchor Loads
Penetrations X7A, X7B, X7C, X7D - Side Bolt Evaluation

Load Condition	Service Level	Node	Tension lb	Shear lb	T allow lb	S allow lb	IR T/Ta+S/Sa
DW+TH+SSE+BREAK (X7D)	D	22	106702	17509	157500	96250	0.859
DW+TH+SSE+BREAK (X7A)	D	30	107227	16683	157500	96250	0.854

Maximum Support Loads
MS Relief Valve Discharge Line Support RV25A-H1 (spring hanger)

Load Condition	Service Level	Node	Max Load lb	Allowable lb	IR Max/Allow	Min Load lb	Allowable lb	IR Allow/Min
DW+TH+SRSS(TSV,SRV,OBE)	B	285	1341	1344	0.998	1162	780	0.671

Table 2
MNGP EPU Main Steam Piping and Support Results Summary

Main Steam Outside Containment - Maximum EPU Results (Highest Interaction Ratio)

Maximum Pipe Stresses

Load Combination	Service Level	Node	Stress psi	Allowable psi	Ratio S/Allow
P+DW	B	X7A	6877	15000	0.46
TH Range	B	TURB	19441	22500	0.86
P+DW+TSV	B	268	12236	18000	0.68
DW+TSV+SRV+SSE	D	268	13795	26325	0.52
HELB DW+TH+OBE	B	TURB	27559	30000	0.92

Maximum Turbine Loads

Load Combination	Service Level	Node	Mx ft-lb	Allowable ft-lb	Ratio Mx/Allow	Mz ft-lb	Allowable ft-lb	Ratio Mz/Allow
DW	B	*	32244	413000	0.078	171446	722000	0.237
DW + TH	B	*	271321	413000	0.657	302310	722000	0.419

* Note: Loads from all turbine nodes were combined

Maximum Support Loads
Main Steam Line Support PS-16, Node 283

Load Condition	Service Level	Component	Max Load lb	Allowable lb	IR Max/Allow
DW+TH+SRSS(TSV,SRV,OBE)	B	Anchor bolt	20026	20731	0.966

Table 3
MNGP EPU BOP Piping and Support Results Summary

Maximum EPU Results (Highest Interaction Ratio)

Extraction Steam

Maximum Pipe Stresses

Load Combination	ANSI B31.1 EQ.	Heater	Stress psi	Allowable psi	Ratio S/Allow
P+DW	11	15A	7944	15000	0.53
P+DW+OCC	12	15A	7967	18000	0.44
P+DW+TH	14	15A	25599	37500	0.68

Note: OCC represents stresses/loadings from the occasional loadings from simplified seismic analysis using Uniform building code (UBC) methodology.

Maximum Support Loads, Support for Heater 14APS-16, Node 283

Load Condition	Pre-EPU IR	EPU % Increase	EPU IR
DW+TH+OCC	0.79	4.64	0.827

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Heater Drains & Vents

Maximum Pipe Stresses

Load Combination	ANSI B31.1 EQ.	Heater	Stress psi	Allowable psi	Ratio S/Allow
P+DW	11	14A-13A	3772	15000	0.25
TH	13	14A-13A	19564	22500	0.87

Maximum Support Loads, Support HDH-73, Feedwater Heater E-13B Dump Line

Load Condition	Pre-EPU Load, lb	Increase %	EPU Load, lb	Support Capacity	EPU IR
DW+TH	1452	51.40%	2198	2200	0.999

Table 3
MNGP EPU BOP Piping and Support Results Summary

Moisture Separator Drain Lines

Maximum Pipe Stresses

Load Combination	ANSI B31.1 EQ.	Tanks	Stress psi	Allowable psi	Ratio S/Allow
P+DW	11	T6A-T6D	4452	15000	0.40*
TH	13	T6A-T6D	18050	22500	0.80

*Reflect results of prompt evaluation of as-built conditions from CARV modifications completed during the 2009 refueling outage.

Maximum Support Loads, Support CDH-64, Moisture Separator 11 Drain Line Support

Load Condition	Pre-EPU IR	EPU % Increase	EPU IR
DW+TH	0.64	30	0.83

Core Spray

Maximum Pipe Stresses

Load Condition	Pre-EPU IR	EPU % Increase	EPU IR
TH	0.91	8.9	0.99

Maximum Support Loads, Support TWH-86, Core Spray Pump Discharge Line Support

Load Condition	Pre-EPU Load, lb	EPU Load, lb	Pre-EPU IR	EPU IR
DW+TH+OBE	1471	1480	0.99	0.996

Table 3
MNGP EPU BOP Piping and Support Results Summary

RCIC Injection

Maximum Pipe Stresses

Load Condition	Pre-EPU IR	EPU % Increase	EPU IR
TH	0.83	8.9	0.904

Maximum Support Loads, Support H-1

Load Condition	Pre-EPU IR	EPU % Increase	EPU IR
DW+TH+OBE	0.997	0	0.997

NOTE: H-1 not affected by EPU increases. Support is remote from temperature increase

HPCI Injection

Maximum Pipe Stresses

Load Condition	Pre-EPU IR	EPU % Increase	EPU IR
DW+TH+OBE	0.78	8.9	0.85

Maximum Support Loads, Support SR-393, Suction Supply Line Support

Load Condition	Pre-EPU IR	EPU % Increase	EPU IR
DW+TH	0.966	0	0.966

NOTE: SR-393 not affected by EPU increases. Support is remote from temperature increase

EMCB RAI No. 14

Verify whether the increased flow rate due to EPU affects the structural analysis (pipe stress and support loads) of only the MS and FW piping.

NSPM RESPONSE

The current design basis includes fluid transient loads only in the Main Steam system. The increased Main Steam flow rate due to EPU is included in the structural analysis (pipe stress and support loads) of the Main Steam piping and attached branch lines. The current licensing basis (refer to USAR Section 12.2.1) does not include flow induced load analyses for the Feedwater piping and none was added for EPU.

EMCB RAI No. 15

The reactor coolant pressure boundary (RCPB) piping systems structural evaluation is contained in Section 2.2.2 of the PUSAR. Please provide structural evaluations for the residual heat removal (RHR) low pressure coolant injection (LPCI) and core spray systems and whether their piping and supports are structurally adequate for the EPU conditions.

NSPM RESPONSE

The only system condition change in either RHR (LPCI) and CS is operation with suppression pool water increased from a peak temperature of 196.7°F (current) to 212°F (EPU). The injection portions of these systems near the reactor were originally analyzed at 570°F and are unaffected by this change. The remainder of the RHR (LPCI) system was originally analyzed at a temperature of 330 ºF representing the shutdown cooling mode of operation, which bound the EPU suppression pool temperature, so the stress analysis results are not changed. The highest stresses for piping and supports in the CS system are summarized in the table below, which indicate the loads are within code allowable values, although very close to the limits. Therefore, the associated piping and supports are structurally adequate for the EPU conditions.

Core Spray

Maximum EPU Results (Highest Interaction Ratio)

Maximum Pipe Stresses

Load Condition	Pre-EPU IR	EPU % Increase	EPU IR
TH	0.91	8.9	0.99

Maximum Support Loads, Support TWH-86, Core Spray Pump Discharge Line Support

Load Condition	Pre-EPU Load, lb	EPU Load, lb	Pre-EPU IR	EPU IR
DW+TH+OBE	1471	1480	0.99	0.996

EMCB RAI No. 16

The PUSAR indicates that "the MS piping pressures and temperatures are not affected by EPU." Please confirm that the main steam piping has no temperature and pressure increases due to the EPU and whether that includes main steam branch piping inside and outside containment including the main steam turbine bypass piping.

NSPM Response

There are no temperature and pressure increases due to the EPU for main steam piping and its branch piping inside and outside containment including the main steam turbine bypass piping.

EMCB RAI NO. 17

Steam flow and feedwater flow will increase as a result of the CPPU implementation. The load due to the TSV fast closure transient is used in the design of the MS piping system. Page 2-31 states that "Due to the magnitude of the TSVC transient load increase [at EPU], the transient event was reanalyzed. The main steam piping was then reanalyzed using this revised load definition."

a) Provide a quantitative summary of the MS and associated piping system evaluation (inside and outside containment), including pipe supports, that contains the increased loading associated with the TSV closure transient at EPU conditions, along with a comparison to the code allowable limits. For piping, include maximum stresses and data at critical locations (i.e. nozzles, penetrations, etc), including fatigue evaluation CUFs, where applicable. For pipe supports, state the method of evaluation for EPU conditions and confirm that the supports on affected piping systems have been evaluated and shown to remain structurally adequate to perform their intended design functions. For non-conforming piping and pipe supports, provide a summary of the modifications required to ensure that piping and pipe supports are structurally adequate to perform their intended design functions and the schedule for completion of these modifications.

b) For FW and condensate, please respond as in part (a) of this RAI.

NSPM RESPONSE

Response to Part a
The Main Steam system piping analysis results, including TSV closure loads are summarized below. The piping system was evaluated (by re-analysis versus scaling) using requirements from the existing code of record. The supports in the Main Steam piping remain adequate to perform their intended design functions. An updated status for PUSAR Table 2.2-2d is provided in response to RAI 12, Part b above. There are no non-conforming pipes or supports requiring modifications on the main steam system.

Main Steam Inside Containment

Maximum EPU Results (Highest Interaction Ratio):

Maximum Pipe Stresses

Load Combination	Service Level	Node	Stress psi	Allowable psi	Ratio S/Allow
P+DW	B	161	7709	15000	0.51
TH Range	B	203	22940	22998	1.00
P+DW+OBE	B	U08	17823	18000	0.99
DW+TSV+SRV+SSE	D	U08	31261	36000	0.87

Note: 1. High Energy Line Breaks locations are not postulated inside containment.
2. Due to the revised analysis of the turbine stop valve closure loads, comparison to pre-EPU values is not meaningful.

Maximum SRV Flange Loads
Inlet Flange

Load Condition	Service Level	Node	Moment ft-lb	Allowable ft-lb	Ratio M/Allow
DW + TH	B	U07	14558	34083	0.427
DW + TH + Level B Dynamic	B	U07	39362	68167	0.577
DW + TH + Level D Dynamic	D	U07	65909	99750	0.661

Outlet Flange

Load Condition	Service Level	Node	Moment ft-lb	Allowable ft-lb	Ratio M/Allow
DW + TH	B	U08	13663	31000	0.441
DW + TH + Level B Dynamic	B	U08	34907	62083	0.562
DW + TH + Level D Dynamic	D	U08	57547	91250	0.631

Maximum RPV Nozzle Loads
RPV Nozzle N-3D

Loads	Service Level	Node	Fx lb	Fy lb	Fz lb	Mx ft-lb	My ft-lb	Mz ft-lb
Maximum Loads	B	101	6667	18555	4979	67422	18193	98764
Allowables	B	101	19392	51712	19392	258562	32320	258562
Maximum/Allowable	B	101	0.344	0.359	0.257	0.261	0.563	0.382

Maximum Flue Head Anchor Loads
Penetrations X7A, X7B, X7C, X7D - Side Bolt Evaluation

Load Condition	Service Level	Node	Tension lb	Shear lb	T allow lb	S allow lb	IR T/Ta+S/Sa
DW+TH+SSE+BREAK (X7D)	D	22	106702	17509	157500	96250	0.859
DW+TH+SSE+BREAK (X7A)	D	30	107227	16683	157500	96250	0.854

Maximum Support Loads
MS Relief Valve Discharge Line Support RV25A-H1 (spring hanger)

Load Condition	Service Level	Node	Max Load lb	Allowable lb	IR Max/Allow	Min Load lb	Allowable lb	IR Allow/Min
DW+TH+ SRSS(TSV,SRV,OBE)	B	285	1341	1344	0.998	1162	780	0.671

Main Steam Outside Containment
Maximum EPU Results (Highest Interaction Ratio):

Maximum Pipe Stresses

Load Combination	Service Level	Node	Stress psi	Allowable psi	Ratio S/Allow
P+DW	B	X7A	6877	15000	0.46
TH Range	B	TURB	19441	22500	0.86
P+DW+TSV	B	268	12236	18000	0.68
DW+TSV+SRV+SSE	D	268	13795	26325	0.52
HELB DW+TH+OBE	B	TURB	27559	30000	0.92

Maximum Turbine Loads

Load Combination	Service Level	Node	Mx ft-lb	Allowable ft-lb	Ratio Mx/Allow	Mz ft-lb	Allowable ft-lb	Ratio Mz/Allow
DW	B	*	32244	413000	0.078	171446	722000	0.237
DW + TH	B	*	271321	413000	0.657	302310	722000	0.419

*Note: Loads from all turbine nodes were combined

Maximum Support Loads
Main Steam Line Support PS-16, Node 283

Load Condition	Service Level	Component	Max Load lb	Allowable lb	IR Max/Allow
DW+TH+SRSS(TSV,SRV,OBE)	B	Anchor bolt	20026	20731	0.966

Response to Part b

The maximum Feedwater system operating temperature is 397.7^{o}F at EPU conditions for the Feedwater piping from the outboard containment isolation valve to the containment and inside containment. This value is bounded by the original analysis temperature of 400^{o}F. The design pressure for this portion of the Feedwater system is unchanged by EPU. Therefore this piping is unaffected by EPU relative to HELB postulation. The current design basis for Feedwater piping analysis does not include fluid transient analysis. The stress analyses for the Feedwater piping from the outboard containment isolation valve to the containment and inside containment are therefore unaffected by EPU.

The feedwater piping and condensate piping from the condensate pump suction to the containment isolation valves will be re-analyzed during the Feedwater and Condensate system modifications (reference response to RAI 7).

EMCB RAI No. 18

In accordance with Section 2.2.2 of the PUSAR, the main steam and associated piping system structural evaluation was performed to justify the operation of these systems at EPU conditions. This evaluation showed that one small bore branch line did not meet the displacement criteria. PUSAR further states that, "Additional detailed analysis will be performed to qualify this line or the piping modified prior to operation at EPU conditions."

a) Provide identification of the small bore branch line (size, system, location, function).

b) Describe the required displacement limits and their bases.

c) Since this piping analysis, with potential piping and or support modifications, is required for EPU, please discuss the reasoning for not including this information in your application. Also, indicate when necessary modifications, as needed, will be completed.

NSPM RESPONSE

a) The branch line is a 1 inch instrument sensing line located inside the primary containment. The line connects one of the differential pressure sensing ports on the D steam line flow restrictor to a containment instrument piping penetration. This line is used for flow sensing in main steam line D and serves a safety related input function to the high flow Group 1 Containment Isolation logic that will automatically isolate the MSIV's in the event of a main steam line break.

b) A differential displacement of 1/16 inch for branch connection points was used as screening criteria in the piping analysis. Those in excess of 1/16 inch were noted as outliers needing further evaluation. The basis for the 1/16 inch criteria is:

1. The 1/16 inch displacement produces an insignificant stress in the branch line which is typically supported by a standard deadweight span (span length from run pipe nozzle connection to first support on the branch).

2. Typical industry practice is to design supports with a gap of 1/8 inch to 1/16 inch. Therefore the displacement due to EPU is absorbed by the support gap and produces minimal stress in the branch line.

3. The 1/16 inch displacement from the run pipe is considered a secondary stress since it is a deflection limiting stress. The piping system allowables for secondary stresses have significant margins beyond the code requirements especially when fatigue cycles are considered.

4. Typical industry practice is to evaluate main pipe run displacements much higher than 1/16 inch. Therefore the relative increase in stresses due to the EPU 1/16 inch increase will not be significant for the branch line.

c) The depth of information provided in the application was developed as described in Section 1 of the PUSAR.
To complete the evaluation of the instrument line noted above, a field verification of the distance between the pipe tap and first support inside the primary containment was required. This verification was completed during the current refuel outage. This small bore branch line meets code allowables, no modification is necessary.

EMCB RAI No. 19

Page 2-31 of the PUSAR states that, “SRV discharge loads are not affected by EPU.” Please clearly present your evaluation of the effects of the safety relief valve (SRV) discharge line and containment loads at EPU conditions, which demonstrates that the current design basis for containment dynamic load definitions for the SRVs are still valid and bound the EPU conditions.

NSPM Response

The evaluation of the effects of containment loads at EPU conditions is presented in PUSAR Section 2.6.1.2. The containment dynamic loads include Loss-of-Coolant Accident (LOCA) loads and SRV loads. The evaluation of the effects of LOCA loads at EPU conditions is presented in PUSAR Section 2.6.1.2.1, and the evaluation of the effects of SRV discharge line loads at EPU conditions is presented in PUSAR Section 2.6.1.2.2. The conclusions of these evaluations are summarized here.

The LOCA dynamic loads include pool swell (PS), condensation oscillation (CO), and chugging (CH). Vent thrust loads, unique to Mark I containment types, are included in the evaluation. The short-term containment response at EPU conditions remain within the range of test conditions used to define the original PS and CO load definitions for Monticello. Vent thrust loads calculated with the short-term containment response at EPU conditions also remain bounded by the plant-specific vent thrust loads calculated during the Mark I Containment Long-Term Program (LTP). The long-term containment response at EPU conditions when chugging would occur are also bounded by the containment conditions used to define the original chugging loads for Monticello. Therefore the current LOCA load definition remains bounding and applicable for Monticello at EPU conditions.

The SRV dynamic loads are influenced by changes in SRV opening setpoint pressure, the mass (length) of SRV discharge line (SRVDL) and suppression pool geometry, including the mass (length) of water in the discharge line at the time of SRV opening. Since the SRV opening setpoint pressure and the SRVDL and pool geometry are not changing for EPU, the SRV dynamic loads for initial SRV actuation are not increased for EPU. The load definition for subsequent SRV actuations is not affected because SRV low-low set logic has been incorporated at Monticello to ensure that subsequent actuations occur only after the water level in the SRVDL has returned to normal. Therefore the current SRV load definition remains bounding and applicable for Monticello at EPU conditions.

EMCB RAI No. 20

Page 2-33 states that:

> "FW piping from the MOVs [downstream from the high pressure heaters] to the condensate pumps will be modified as a result of the replacement of the feedwater and condensate pumps, and will be qualified for full EPU operation as part of the modification. The current piping and associated components are adequate for operation within the capability of the existing feedwater and condensate pumps."

Page 2-61 indicates that:

> "BOP FW from the condensate pumps to the first isolation valves (IV) (outside containment) "will be analyzed and qualified with the FW and Condensate pump modifications prior to operation at EPU conditions."

a) In addition to the minimum flow line modifications for EPU FW and condensate pumps (identified in Enclosure 8, Planned Modifications), what other piping modifications are anticipated?

b) Indicate whether piping (including supports) analysis at the EPU conditions of the above mentioned FW and condensate piping modifications (including minimum flow lines) has been completed and discuss the analysis results.

c) Provide an explanation whether any transients are applicable in the sections of piping mentioned above (including pump min flow lines) and evaluate their affects with regard to structural integrity of the proposed modifications of piping, pipe components and pipe supports.

NSPM RESPONSE

a) Details of the modifications to the condensate and FW system are not yet finalized. The design goal is to maintain stresses in the existing condensate and FW piping within code allowable limits of ANSI-B31.1-1977, including Winter 1978 Addenda (reference response to RAI 7)

b) The piping analysis for the FW and condensate piping modifications has not been completed.

c) There are no fluid transients applicable to this piping. The piping is non-safety related/seismic Class II piping. It is analyzed for deadweight, pressure and thermal stresses. A portion of the piping from the 13A &B heaters to the FW pumps and from the FW pumps to the 15A&B heaters is analyzed to Class I seismic requirements. These stresses are imposed so that a Class II pipe rupture need not be postulated. (Reference response to RAI 7)

EMCB RAI No. 21

PUSAR, on page 2-33, to makes the following statement with regard to FW pipe stress evaluation:

"A review of the small increases in pressure, temperature and flow associated with EPU indicates that the EPU temperature, pressure and flow conditions are bounded by the existing analyses. The original design analyses have sufficient design margin between calculated stresses and ANSI-B31.1-1977, including Winter 1978 Addenda Code allowable limits to justify operation at EPU conditions."

Explain the small increases in FW flow between OLTP and EPU and between CLTP and EPU that are bounded by the existing analyses, and whether the existing analyses contain flow induced loads at the OLTP or CLTP.

NSPM RESPONSE

The portion of the Feedwater system evaluated in the PUSAR is from the motor operated (MO) valve downstream of high pressure heaters through the containment to the RPV. In this portion of the system, Feedwater flow increases from 7.235E+6 lbm/hr at CLTP to 8.497E+6 lbm/hr at EPU. The temperature and pressure used in the CLTP stress analyses bound EPU conditions. The current licensing basis (refer to USAR Sections 12.2.1.10 and 12.2.2) does not include flow induced load analyses for the Feedwater piping and none was added for EPU. The remainder of the Feedwater piping and Condensate piping includes more significant changes and will be evaluated during the modification design process of the Feedwater and Condensate pump and heater replacement modifications (reference response to RAI 7).

EMCB RAI No. 22

PUSAR, on page 2-33, makes the following statement with regards to the FW pipe support evaluation:

"The FW system was evaluated for the effects of seismic, deadweight and thermal expansion displacements on the piping snubbers, hangers, and struts. A review of the increases in temperature and FW flow associated with EPU indicates that the EPU conditions are bounded by the existing analyses."

Provide a discussion which shows that the FW flow induced loads on pipe supports in the existing analysis bound the EPU flow induced loads.

NSPM RESPONSE

The portion of the Feedwater system evaluated in the PUSAR is from the motor operated (MO) valve downstream of high pressure heaters through the containment to the RPV (reference response to RAI 21). The design and licensing basis (refer to USAR Sections 12.2.1.10 and 12.2.2) for Feedwater piping analysis do not include consideration of flow induced transient loads. No flow induced transient analysis was performed for Feedwater piping. Flow induced vibration is evaluated and will be monitored during power ascension as discussed in EPU LAR Enclosures 9 and 10. The start up and power ascension vibration monitoring program will demonstrate that steady state flow induced vibration at EPU conditions remains within pre-established acceptance limits.

EMCB RAI No. 23

a) Discuss whether there is any piping analysis, in the current design basis of the plant, that contains stratification or discuss whether there is any CLTP stratification monitoring currently in place.

b) If a stratification phenomenon currently exists, explain how these stratification locations have been evaluated at EPU conditions and provide a summary of their evaluation results.

NSPM RESPONSE

MNGP piping analyses do not include consideration of thermal stratification. To validate this, MNGP installed thermocouples on top and bottom of the horizontal Feedwater lines at elevation 952'-10" in the drywell and monitored conditions during startup and shutdown to verify that no global thermal stratification occurs on Feedwater lines due to interaction with RWCU. This was monitored over several start ups and shutdowns with very similar results each time. The results did not show any sign of significant stratification. With relatively minor changes in temperatures in the Feedwater and RWCU systems, it is expected that no global thermal stratification will occur in these lines at EPU.

EMCB RAI No. 24

Consider the two statements below:

LAR Enclosure 10, on page 3 of 16, states the following:

> "If the vibration level in the main piping in these systems [(FW and MS)] is greater than 50% of the acceptance criteria, then an engineering evaluation of the small bore piping will be performed to ensure that the steady state stresses are within the endurance limit."

In response to NRC staff RAI, CLTR for the EPU generic evaluation states that:

"[T]ypically the measured piping vibration levels of the MS and FW piping are only a few percent of [the acceptance] criteria. Hence, the vibration levels of the large bore piping are small and therefore the vibration levels of components and branch piping attached to the large bore piping are not of concern. However, if during testing, the vibration levels of the large bore MS and FW piping are found to be significant, [[say 50% or higher of their acceptance criteria,{3}]] then the attached components and branch piping connections will have a higher probability of fatigue failure relative to operation at the original power level. Hence when the measured MSL or FW large bore piping vibration levels reach [[50% of{3}]] their acceptance criteria, the attached branch piping connections will be further evaluated."

EMCB RAI 24 a)
Please revise the statement of Enclosure 10 of the LAR to be in accordance with the generic CLTR evaluation, in that if the vibration levels of the main piping reach 50% or higher, an engineering evaluation of all attached branch piping, not just for the small bore, will be performed to ensure that the steady state stresses are within the endurance limits. As this was the intention of the CLTR statement.

NSPM RESPONSE

a) The subject statement of Enclosure 10 is revised as follows.

"If the tested vibration level in the main piping in these systems (FW and MS) is greater than 50% of the acceptance criteria, then an engineering evaluation of the attached branch piping connections will be performed to ensure that the steady state stresses are within the endurance limits."

EMCB RAI 24 b)

It appears that the 50% was based on the CLTR statement that, "measured piping vibration levels of the MS and FW piping are only a few percent of [the acceptance] criteria." However, in the Monticello case, from readings taken at 100% CLTP, vibration resulted in levels well above just a few percent of the acceptance criteria. At CLTP, 10 locations came in at above 20% of the acceptance criteria for FW and MS. Inside containment, the maximums were 14% and 32% of the acceptance criteria for FW and MS, respectively. Outside containment, the maximums were 43% and 34% of the acceptance criteria for FW and MS, respectively. Using the EPU expected vibration increase of 32%, the CLTP values of 14, 20, 32, 34 and 43 percent of the acceptance criteria are projected for the EPU to be 18, 26, 42, 45 and 57 percent of the acceptance criteria, respectively.

EMCB RAI 24 (b) 1

.Using the 50% or higher criterion, one location has been predicted to be 57% of the acceptance criterion. Please discuss whether evaluations have been performed for branch lines in the vicinity of this location? Provide a discussion of the evaluation results.

NSPM RESPONSE

An evaluation of the branch lines in the vicinity of this location has not been performed. The 57% projection is conservatively determined, such that the actual testing results may not exceed the 50% criterion. The piping located outside containment was designed to ASME B31.1, yet the acceptance criteria was developed using a conservative approach. If the actual results do exceed the 50% criterion, the response to part a) will apply.

EMCB RAI 24 (b) 2

Provide a basis for justification that the 50% criterion, which the CLTR recommends for cases where piping vibration levels are only a few percent of the acceptance criteria, is applicable for Monticello, where the vibration levels even at CLTP have reached well above a few percent of the acceptance criteria, as shown above.

NSPM RESPONSE

The acceptance criteria were conservatively developed so that even if some of the vibration levels at CLTP are more than a few percent of the acceptance criteria it is not an indication that the piping may have a fatigue failure of a branch line.

The piping vibration levels that have reached at least 30% of the acceptance criteria are mainly located in the steam tunnel and turbine building. This piping is designed to ASME B31.1. The seismic analysis used to determine/locate the supports for these piping systems is typically a static analysis. The acceptance criteria for the MS and FW piping were developed by conservatively applying flat dynamic spectra in all three orthogonal directions. The flat dynamic spectra used to develop the acceptance criteria are extremely conservative since the same magnitude is applied to all frequencies. The steam tunnel and turbine building piping do not have many supports. Due to the low number of supports, there are numerous low natural frequencies that normally would not be subjected to the same magnitude of input than would be at higher natural frequencies.

The input used to develop the acceptance criteria is very conservative because the acceptance criteria were developed prior to collecting any steady state vibration data. If the steam tunnel and turbine building acceptance criteria were revised to reflect more realistic magnitudes, the margin would be greater.

EMCB RAI No. 25

With regard to the reactor pressure vessel (RPV) evaluation for EPU, Page 2-45 of the PUSAR states that:

"The Top Head and Cylindrical Shell and the Stabilizer Bracket were not evaluated for fatigue at the time that the OLTP evaluation was performed, and have not been evaluated for EPU."

Monticello USAR Rev 24, Section 4.2.1 states that:

"[T]he reactor vessel was also designed for the transients which could occur during the design [ ] life. The reactor vessel was analyzed for the cycles listed in Table 4.2-1."

Provide an evaluation which shows that the RPV top head and cylindrical shell and the RPV supports will be structurally adequate at EPU conditions for the renewed plant life.

NSPM RESPONSE

The purchase specification for the Monticello RPV defines the regions and components of the vessel that are to be analyzed. These regions/components include the head closure, bottom head, shell adjacent to the reactor core, reactor vessel supports and stabilizers, supports for reactor vessel internals, control rod drive penetration, feedwater nozzle, poison nozzle, emergency core cooling nozzles, drive system return nozzle, and all nozzles 10" or larger in size.

A summary stress report was generated at the time of vessel fabrication for the vessel shell and top head. [[

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The summary report for the shell and top head includes a summary of all major discontinuities in the shell and top head. These include the dryer hold down bracket, guide rod bracket, steam outlet nozzle, steam dryer support bracket, stabilizer bracket, feedwater bracket, feedwater nozzle, core spray bracket, core spray nozzle, top and bottom insulation brackets, recirculation inlet and outlet nozzles, shroud support, support skirt and knuckle, and the jet pump riser pad. In addition, refueling bellows reactions were considered.

Duty cycles specific to the shell and top head are not defined in the purchase specification; [[

]] the shell and top head are considered structurally adequate for operation at EPU operating conditions for the license period of 60 years.

EMCB RAI No 26

Table 2.2-4 of the PUSAR shows that the fatigue CUFs for the recirculation (RRS) inlet nozzle (Ri) and FW Nozzle significantly increased for EPU by 146% for Ri and 47% for FW nozzle, placing the FW nozzle within approx 8.6% of its limit. Provide an explanation for these significant EPU CUFs increases, confirm that these CUFs are to the end of renewal life and assure that all required transients at EPU conditions have been properly included for these fatigue evaluations.

NSPM RESPONSE

An extensive review of the component history was performed for these components. The key items of the review that resulted in the CUF changes due to EPU are shown below.

[[

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Legend: $3S_m$ = 3 times S_m, where S_m = Design Stress Intensity

								]]

Legend:
S_n =maximum primary plus secondary stress intensity
K_t = stress concentration factor
K_e = elastic-plastic stress concentration factor
S_{alt} = amplitude (half-range) of stress fluctuation
N = number of allowable cycles
n = number of required cycles
u = usage factor for the given stress
U = cumulative usage factor

[[

						]]

Legend:
S_n =maximum primary plus secondary stress intensity
F_{th} = thermal peak stress
K_t = stress concentration factor
S_p = peak stress
S_{alt} = amplitude (half-range) of stress fluctuation
N = number of allowable cycles
n = number of required cycles
U = cumulative usage factor

[[

						]]

Legend: S_n =maximum primary plus secondary stress intensity
F_{th} = thermal peak stress
K_t = stress concentration factor
S_p = peak stress
S_{alt} = amplitude (half-range) of stress fluctuation
N = number of allowable cycles
n = number of required cycles
U = cumulative usage factor

EMCB RAI No. 27

In Table 2.2-9 of the PUSAR, some of the locations are shown with "--". Please explain what is meant by this designation.

NSPM RESPONSE

The '--' designation is used in the Location column under the EPU portion of the Table and means not applicable for Items 8 and 16. For Items 6a, 6b, 7, 12 14, and 15, the '--' designation means the specified locations of the stresses for the components in question are not known.

EMCB RAI No. 28

In Section 2.2.2 of the PUSAR, it is stated that, "The effects of FIV induced stresses at EPU conditions on safety-related thermowells in the MS and FW system and the sample probe in the FW system were evaluated" and indicates that they remain acceptable under EPU conditions (see page 2-28 of the PUSAR). However, Enclosure 8, page 5 of 9 states, "Replace or remove the thermowells in main steam piping to insure appropriate margin for flow induced vibration." Provide a quantitative summary of the evaluation that supports the acceptability of the thermowells and sample probes in the MS, FW and related piping systems. Identify nonconforming component(s) and provide description of their modification(s).

NSPM RESPONSE

A quantitative summary of the evaluation results are provided in the table below:

Component	CLTP			EPU		
	Zero-to-peak stress (psi)	fs/fn	Reduced Velocity	Zero-to-peak stress (psi)	fs/fn	Reduced Velocity
FW Thermowells	2683	0.57	1.30	4536	0.67	1.53
FW Sample Probes	1627	0.31	0.70	2332	0.36	0.82
MS Thermowells	1308	0.70	2.32	2809	0.82	2.73

The stress results were compared to the 13,600 psi endurance limit for all the materials of the probes and thermowells. At EPU conditions, all of the stress values are below this endurance limit and thus the thermowells and sample probes are structurally adequate. However, it is desired to reduce the ratio of the vortex shedding frequency to the natural frequency of the MS thermowells (TE 2-127A & B) to the CLTP value to minimize the potential of the system jumping into resonance. Reducing the length of the thermowells by 10% will accomplish this goal. Currently two options are being evaluated; either replace the thermowells with shorter ones or remove them altogether. Final resolution of this issue is now scheduled for the 2011 refuel outage.

EMCB RAI No. 29

Page 2-59 of the PUSAR states that:

"The temperatures, accident radiation level, and the normal radiation level increase due to EPU. These effects are not considered to have an adverse effect on the functional capability of nonmetallic components in the mechanical equipment both inside and outside containment."

Please provide a justification that the radiation due to the EPU is not higher than the radiation damage threshold of the non-metallic parts of the resilient seated check valves, hydraulic snubbers and flex joint bellows affected by the EPU.

NSPM RESPONSE

The predicted dose increase due to EPU operation was determined for all plant general areas. The prediction is that the dose will increase slightly. MNGP will perform plant radiation surveys during power ascension testing and at EPU (power operation and post shutdown) to confirm predicted radiation dose rates.

MNGP has active and formal programs in place to properly manage the slight increase in radiation expected for EPU. The subject components are procured and designed for the applicable service environments in accordance the requirements of the Quality Assurance Program. This program includes requirements to assure that plant equipment is suitable for the intended service, and is of acceptable quality consistent with their effect on safety.

The MNGP Check Valve Program closely monitors valve reliability. The program monitors check valve maintenance history and check valve failures. The check valves with non-metallic seals are included in the program. Valves with non-metallic seats receive regular maintenance including inspection and bench testing. The valves are functionally evaluated during maintenance and replaced if necessary. The O-rings and seals are typically replaced regardless of condition. This program has provided reliable check valve performance to date at Monticello, and the slight increase in radiation due to EPU is not expected to have an adverse effect on continued reliability.

Like the check valve program, the MNGP Snubber Program closely monitors snubber reliability. The program monitors maintenance history and snubber failures. The in-service requirements are delineated by Section 3.4.3 of MNGP Technical Requirements Manual. The installation and maintenance records for each safety-related snubber are reviewed at least once every 24 months to verify that the indicated service life will not be exceeded prior to the next scheduled service life review. The service life of a snubber is evaluated via manufacturer input and thorough consideration of the snubber service conditions and associated installation and maintenance records (newly installed snubber, seal replacement, spring replacement, in high radiation, in high temperature area, etc.). The requirement to monitor the snubber service life is included to ensure that the snubbers periodically undergo a performance evaluation in view of their age and operating conditions. These records provide statistical bases for future consideration of snubber service life. In addition, to address seal failures specifically, Monticello has assigned a maximum service life of 10 years for all hydraulic snubbers regardless of installed location in the plant. If degradation or damage is detected, the number overhauled will be adjusted. This was noted in an NSP letter (L. O. Mayer) to the AEC (J. F. O'Leary) dated October 1, 1974.

This program has provided reliable snubber performance to date at Monticello, and the slight increase in radiation due to EPU is not expected to have an adverse effect on continued reliability.

A database search of the MNGP plant equipment did not identify expansion bellows with elastomer components. Regarding expansion joints, the plant systems with rubber expansion joints were identified as part of the MNGP License Renewal Program. These components are not located in safety related systems (e.g. Condenser, Service Water System). The program determined that changes to material properties for rubber required a source strength of 10E7 Rads, and concluded that these components were not susceptible to hardening and loss of strength caused by radiation as there is significant margin to this value. This margin exceeds that which may occur due to the conservative 13% increase in radiation expected for EPU. In addition, the systems that were identified to contain rubber expansion joints are within the scope of the MNGP Maintenance Rule Program. The program monitors system reliability and a significant increase in failures of rubber expansion joints for these systems has not been noted. The slight increase in radiation for EPU is not expected to have an adverse effect on the reliability of systems containing rubber expansion joints.

EMCB RAI No. 30

Page 2-59 of the PUSAR states that:
"The Monticello design and licensing bases do not require a formal mechanical EQ program like the EQ program applied to electrical equipment."

What program used at Monticello establishes the capability of active safety-related mechanical equipment and their components to perform their required safety function for the life of the plant during postulated normal and accident conditions?

NSPM RESPONSE

Monticello does not have a formal mechanical EQ program. The remainder of the PUSAR paragraph cited above describes the programs that are in place at Monticello. The key elements are design control, testing/preventive maintenance and equipment monitoring in accordance with the maintenance rule. A key element of the maintenance rule is to also incorporate industry-wide operating experience into the program. The integrated effect of these elements provides reasonable assurance that important systems, structures and components will be capable of fulfilling their intended functions.

ENCLOSURE 4

United States Atomic Energy Commission - Safety Evaluation by the Directorate of Licensing, Docket No. 50-263, Monticello Nuclear Generating Plant - "Analysis of the Consequences of High Energy Piping Failures Outside Containment", July 29, 1974

UNITED STATES
ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

AUG 0 5 1974

NUCLEAR SUPPORT SERVICES DEPT.

DISCARD

Forward: to Monticello
SAC 8-5-74

July 29, 1974

Docket No. 50-263

Northern States Power Company
ATTN: Mr. L. O. Mayer, Director of
Nuclear Support Services
414 Nicollet Mall
Minneapolis, Minnesota 55401

Gentlemen:

Your letter dated September 7, 1973, submitted a report on Postulated Pipe Failures Outside Containment dated August, 1973 for the Monticello Nuclear Generating Plant as required by our letter of December 18, 1972. Our letters to you dated December 18, 1972, and January 16, 1973, contained criteria by which a postulated rupture in any high energy fluid piping outside the primary containment was to be evaluated to assure safe plant shutdown capability. On February 5, 1973, a meeting was held to discuss these criteria and the results of your preliminary study of the Monticello Nuclear Generating Plant. Upon completion of our initial review of your report, further clarification of the analysis was requested in our letter of January 18, 1974. Your response to this letter was dated March 8, 1974.

Based upon our review of the information you have submitted on the Monticello Nuclear Generating Plant, we have determined that after completion of the proposed modifications, the plant would withstand the consequences of postulated ruptures in high energy fluid piping outside the containment without loss of the capability to initiate and maintain the plant in a safe shutdown condition. You have informed us that these modifications were completed during the spring 1974 refueling outage. Therefore, we conclude that the Monticello Nuclear Generating Plant now complies with the requirements of our December 18, 1972 letter. A copy of our related Safety Evaluation is enclosed.

Sincerely,

Karl R. Goller

Karl R. Goller, Assistant Director
for Operating Reactors
Directorate of Licensing

Enclosure:
Safety Evaluation

cc on next page

1229

cc w/enclosure:
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UNITED STATES ATOMIC ENERGY COMMISSION

SAFETY EVALUATION BY THE DIRECTORATE OF LICENSING

DOCKET NO. 50-263

MONTICELLO NUCLEAR GENERATING PLANT

ANALYSIS OF THE CONSEQUENCES OF HIGH ENERGY PIPING FAILURES OUTSIDE CONTAINMENT

INTRODUCTION

On December 18, 1972, and January 16, 1973, the Atomic Energy Commission's Regulatory staff sent letters to Northern States Power Company requesting a detailed design evaluation to substantiate that the design of the Monticello Nuclear Generating Plant is adequate to withstand the effects of a postulated rupture in any high energy fluid piping system outside the primary containment, including the double-ended rupture of the largest line in the main steam and feedwater system. It was further requested that if the results of the evaluation indicated that changes in the design were necessary to assure safe plant shutdown, information on these design changes and plant modifications would be required. Criteria for conducting this evaluation were included in the letters. A meeting was held on February 5, 1973, to discuss the information already available on the Monticello Plant design concerning postulated pipe ruptures, to discuss the criteria, and to assess those areas where additional information was required. In response to our letters, a report concerning postulated high energy pipe ruptures outside containment was filed by Northern States Power Company with letter dated September 7, 1973. A subsequent letter from Northern States Power Company dated March 8, 1974, answered additional questions in a letter from the staff dated January 18, 1974.

EVALUATION

Criteria

A summary of the criteria and requirements included in our letter of December 18, 1972, is set forth below:

a. Protection of equipment and structures necessary to shutdown the reactor and maintain it in a safe shutdown condition, assuming a concurrent and unrelated single active failure of protected equipment, should be provided from all effects resulting from ruptures in pipes carrying high energy fluid, where the temperature and pressure conditions of the fluid exceed 200°F and 275 psig, respectively, up to and including a double-ended rupture of such pipes. Breaks should be assumed to occur in those locations specified in the "pipe whip criteria". The rupture effects to be considered include pipe whip, structural (including the effects of jet impingement), and environmental.

b. In addition, protection of equipment and structures necessary to shutdown the reactor and maintain it in a safe shutdown condition, assuming a concurrent and unrelated single active failure of protected equipment, should be provided from the environmental and structural effects (including the effects of jet impingement) resulting from a single open crack at the most adverse location in pipes carrying fluid routed in the vicinity of this equipment. The size of the cracks should be assumed to be 1/2 the pipe diameter in length and 1/2 the wall thickness in width.

High Energy Systems

Our evaluation included the following piping systems containing High Energy fluids:

Main, Extraction, and Auxiliary Steam Systems
Feedwater System
Condensate System
Reactor Core Isolation Cooling System (RCIC)
High Pressure Coolant Injection System (HPCI)
Reactor Water Cleanup System (RWCU)
Residual Heat Removal System (RHR)
Sample Lines (Environmental Effects Only)

Areas or Systems Affected by High Energy Pipe Breaks

An evaluation was conducted of the effects of high energy pipe breaks on the following systems, components, and structures, which would be necessary (in various combinations, depending on the effects of the break) to safely shutdown, cooldown, and maintain cold shutdown conditions:

a. General

1. Control Room
2. Control and Instrument Cables and Tunnels
3. Electrical Distribution System
4. Emergency dc Power Supply (batteries)
5. Emergency ac Power Supply (diesels)
6. Heating and Ventilation Systems (needed for long-term occupancy to maintain the reactor in safe shutdown condition)

b. Reactor Control Systems and associated instrumentation

c. Cooling and Service Water Systems

d. ECCS components

Specific Areas of Concern

The applicant has provided the results of his examination of all postulated safety related high energy line break locations and evaluated the break consequences. We have reviewed all of this information, including the following specific areas of concern where the potential consequences might be severe or where specific corrective action would further assure safe cold shutdown of the plant.

a. Compartment Pressurization

Large pipe breaks, including the double-ended rupture of the largest pipes in a system, and small leakage cracks up to the design basis size have been considered for the main steam tunnel, the turbine building, the ECCS rooms, and the valve compartments.

In the condenser compartment, a failure of a main steam line would pressurize the condenser compartment to 1.4 psig maximum with a vent area of approximately 500 ft^2. The vent area is sufficient to prevent damage or loss of safety equipment and to keep the peak pressure well below the 8.4 psig design.

In the main steam tunnel, the effects of a main steam line break were considered as the design cases. The resultant pressure was calculated to increase to 12.2 psig.

The vent area of 180 ft^2 for the main steam chase is provided by ventilation ducts, doorways, and blowout panels between the tunnel and above the turbine operating deck. The vent area is sufficient to keep the peak pressure below the design of 13.4 psig.

A failure of the Reactor Core Isolation Cooling System (RCIC) steam line could result in the loss of one emergency service water line. Even if a single failure disabled the redundant line, emergency measures of manual connection could be accomplished and allow cooling water to required components in time to enable a safe shutdown of the plant.

An HPCI steam line failure in the HPCI compartment could result in discharge of steam until automatic isolation is achieved. A maximum pressure occurring in the compartment has been calculated to reach 0.9 psig which is below the structural capabilities of approximately 2.0 psig.

A postulated RWCU high energy line failure in the cleanup system pump or heat exchanger compartments results in a single-ended piping failure until isolation is achieved. A check valve in the upstream connection into the feedwater piping would prevent extensive backflow. The calculated pressure resulting from a pipe failure would be 0.6 psig in the heat exchanger room, and 0.2 psig in the pump compartment. The minimum design capacities of these compartments are 16.0 psi for both the heat exchanger and pump compartments.

b. Pipe Whip

The reactor and turbine building areas were considered for the effect of pipe whip and jet impingement from the main steam feedwater and condensate lines. The steam tunnel has been designed with thick reinforced concrete capable of withstanding large static and dynamic loads. The reinforced concrete steam tunnel in which the main steam and feedwater lines are routed from the primary containment to the turbine room is subjected only to the loads of the piping and a live load from the floor on top of the tunnel roof. A whipping main steam or feedwater line in the main steam tunnel could cause rupture of the HPCI, RCIC turbine steam inlet lines and the RWCU line. However, loss of these lines would not impair safe shutdown of the plant.

A jet impingement from a broken feedwater line in the area of vital motor control centers (MCC) could cause some loss to redundant safeguards equipment by failure of the mezzanine floor. The addition of additional piping restraints in this area will reduce the forces which impinge on the mezzanine floor and thereby protect the MCC.

Rupture of condensate piping would not cause a hazard to safeguards equipment and would only cause minor flooding. The high energy line to the HPCI turbine inlet is routed above the torus, therefore rupture of this line may deform the torus. However, the torus would not rupture and would remain functional.

Other high energy lines such as the sample lines and reactor water cleanup lines are located such that their rupture would not cause damage to the torus. A whip of either the RCIC or HPCI steam line outside the torus compartment could damage system isolation valves on the HPCI or RCIC lines. However, the resultant damage would not impair safe shutdown of the unit.

c. Control Room Habitability

The main control room is physically isolated from all high energy lines. Neither the control room equipment nor its ventilation system will be affected by environmental effects caused by a rupture of a high energy line.

d. Environmental Effects

Components and equipment were analyzed and checked for possible adverse environmental effects which could be caused by the rupture of a high energy line. Adverse temperature, pressure, and humidity were the parameters which were used in the evaluation of safety related equipment. We have reviewed the licensee assessment of the consequences of environmental effects on safety related equipment. We find that safety related equipment has been designed to limits in excess of postulated conditions which could arise from the rupture of a high energy line.

Modifications

Modifications to the existing facility are currently being undertaken by Northern States in order to assure that the design will have adequate safety margins in the event of a high energy line rupture outside the containment. These modifications are to be complete prior to restart of the plant following the current refueling outage, currently scheduled to end on or about May 15, 1974.

Additional piping restraints are being installed in the areas of the mezzanine floor near the MCC where a jet impingement could cause a loss of redundant safeguards equipment. With the addition of the pipe restraints, the possibility of jet impingement and pipe whip has been reduced and provides reasonable assurance that redundant safeguards equipment will not be lost.

CONCLUSIONS

On the basis of this review of the information submitted to us and on discussions with Northern States Power, we find that their assessment of the consequences of high energy line failures outside containment is acceptable. Some modifications are necessary. We have concluded that the potential consequences of these postulated high energy pipe failures, following the modifications, will not prevent the capability to achieve safe cold shutdown conditions consistent with the single failure and redundancy requirements as described in our letter of December 18, 1972.

The licensee has stated that the modifications were completed prior to returning to operation from the spring 1974 refueling outage. With the completion of these modifications, there is reasonable assurance that the health and safety of the public will not be endangered by continued operation.

James C. Snell
James C. Snell
Operating Reactors Branch #2
Directorate of Licensing

Dennis L. Ziemann
Dennis L. Ziemann, Chief
Operating Reactors Branch #2
Directorate of Licensing

Date: July 29, 1974